U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - None

I. AREAS OF REVIEW

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel. General Design Criteria 1, 2, 4 and 10 and 10 CFR Part 50, §50.55a require that structures and components important to safety shall be constructed and tested to quality standard commensurate with the importance of the safety functions to be performed, and designed with appropriate margins to withstand effects of anticipated normal plant operational occurrences; natural phenomena such as earthquakes; postulated accidents including loss-of-coolant accidents (LOCA), and from events and conditions outside the nuclear power unit.

For the purpose of this standard review plan section, the term "reactor internals" includes core support structures and other internal structures and refers to all structural and mechanical elements inside the reactor pressure vessel with the exception of the following:

1. Reactor fuel elements, the reactivity control elements out to the coupling interfaces with the drive units (the fuel system design is covered in Standard Review Plan (SRP) Section 4.2, but the structural aspects of reactor fuel assemblies are reviewed with the reactor internals).

2. Control rod drive elements (the drive elements inside the guide tubes are covered in SRP Section 3.9.4, but the guide tubes are reviewed with the reactor internals).

3. In-core instrumentation (in-core instrumentation support structures are reviewed with the reactor internals).

The staff review includes the following specific areas:

a. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems should be presented, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

b. The loading conditions that provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events should be specified. All combinations of design and service loadings should be listed (e.g., operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the reactor internals.

c. The design bases for the mechanical design of the reactor vessel internals should be presented including allowable limits such as maximum allowable stresses; stability under dynamic loads; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and hold-down). Details of dynamic analyses, input forcing functions, and response to loadings are discussed in SRP Section 3.9.2.

d. Each combination of design and service loadings should be categorized with respect to the allowable design or service limits (defined in the ASME Code and SRP Section 3.9.5, Reference 5 and 7), and the associated stress intensity or deformation limits should be stipulated. Design or service loadings should include safe shutdown earthquake (SSE) and operating basis earthquake (OBE) loads as appropriate.

In addition, the MEB will coordinate other branches' evaluations that interface with the overall review of the reactor internals as follows:

The Core Performance Branch (CPB) will verify fuel system design, including fuel behavior effects to reactor core design under various normal and accident operating conditions in SRP Section 4.2. The Materials Engineering Branch (MTEB) will review material aspects of reactor internals in SRP Section 4.5.2.

For those areas of review identified above as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

MEB acceptance criteria are based on meeting the requirements of the following regulations:

1. General Design Criterion 1 and 10 CFR Part 50, §50.55a, as it relates to reactor internals, requires that the reactor internals shall be designed to quality standards commensurate with the importance of the safety functions to be performed.

2. General Design Criterion 2, as it relates to reactor internals, requires that the reactor internals shall be designed to withstand the effects of earthquakes without loss of capability to perform its safety functions.

3. General Design Criterion 4, as it relates to reactor internals, requires that reactor internals shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA.

4. General Design Criteria 10, as it relates to reactor internals, requires that reactor internals shall be designed with adequate margins to assure specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences.

Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

a. Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in SRP Section 3.9.3 (Ref. 7).

b. The design and construction of the core support structures should conform to the requirements of Subsection NG, "Core Support Structures," of the ASME Code (Ref. 5), and SRP Section 3.9.3.

c. The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

d. Deformation limits for reactor internals should be established by the applicant and presented in his safety analysis report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are discussed in SRP Section 3.9.2.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

The configuration and general arrangement of all mechanical and structural internal elements covered by this SRP section are reviewed and compared to those of previously licensed similar plants. Any significant changes in design are noted and the applicant is asked to verify that these changes do not affect the flow-induced vibration test results required by SRP Section 3.9.2.

With respect to the design and analysis of reactor internals, a statement by the applicant that they are designed in accordance with Subsection NG and SRP Section 3.9.3, "Core Support Structures," of Reference 5 and 7 is acceptable. In lieu of such a commitment, the reviewer must determine that the design and analysis of these components are consistent with the requirements discussed in subsection II, above. This is accomplished by requiring that the applicant describe the design procedures and criteria used in the design of these components. This includes a list of the design and service stress limits used for all of the applicable loading conditions.

The deformation limits specified for these components are reviewed to verify that the applicant has stated that these deflections will not interfere with the functioning of related components, e.g., control rods and standby cooling systems, and that the stresses associated with these displacements are less than the specified limits for the core support structures.

At the operating license stage, the calculated stresses and deformations are reviewed to determine that they do not exceed the specified limits.

Any deviations that have not been adequately justified are identified and findings to that effect are transmitted to the applicant with a request for conformance with the requirements discussed in subsection II, above, or additional technical justification.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this SRP section and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the design of reactor internals is acceptable and meets the requirements of General Design Criteria 1, 2, 4, and 10 and 10 CFR Part 50, §50.55a. This conclusion is based on the following:
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> 1. The applicant has met the requirements of GDC 1 and 10 CFR Part 50, §50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.
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> 2. The applicant has met the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated loss-of-coolant accidents with sufficient margin to assure that capability to perform its safety functions is maintained and the specified acceptable fuel design limits are not exceeded.
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>   The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."
2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena."
3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."
4. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."
5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.
6. Standard Review Plan Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."
7. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."